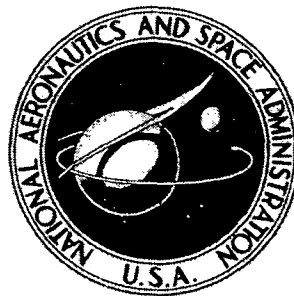


**NASA TECHNICAL
MEMORANDUM**



NASA TM X-1802

NASA TM X-1802

**A NEW METHOD FOR TREATMENT OF
UNCERTAINTIES IN NUCLEAR REACTOR
HEAT TRANSFER CALCULATIONS**

by

Harry J. Reilly

Lewis Research Center

and

John D. Hansell and George L. Heath

University of Toledo

A NEW METHOD FOR TREATMENT OF UNCERTAINTIES IN
NUCLEAR REACTOR HEAT TRANSFER CALCULATIONS

By Harry J. Reilly

Lewis Research Center
Cleveland, Ohio

and

John D. Hansell and George L. Heath

University of Toledo
Toledo, Ohio

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

For sale by the Clearinghouse for Federal Scientific and Technical Information
Springfield, Virginia 22151 - CFSTI price \$3.00

ABSTRACT

A method for performing probability calculations for fuel clad surface temperatures of nuclear reactors is presented. The method properly treats factors which vary systematically over the whole core, factors which vary over a plate, and those that vary at a spot. One method of choosing spot size for plate-type reactors is shown and a more general method is discussed. The method is not restricted to normal distributions. The calculated value for probability of success, that is, surface temperature not exceeding coolant saturation temperature, is identified as strictly applicable only to the instant of reactor startup. Also the effect of continued operation is discussed. It is observed that the probability of success may be more nearly equal to values obtained by the Deterministic Method than to those obtained by the Statistical Method.

A NEW METHOD FOR TREATMENT OF UNCERTAINTIES IN NUCLEAR REACTOR HEAT TRANSFER CALCULATIONS

by Harry J. Reilly, John D. Hansell,* and George L. Heath*

Lewis Research Center

SUMMARY

A method of performing probability calculations for nuclear reactor surface temperatures has been devised. The method gives consideration to the fact that some uncertainties may vary systematically over all the reactor or over some parts of it. The method does not depend on assumption of any particular form of the probability distributions.

The method was used to do an example calculation for an MTR-type test reactor with plate-type fuel elements. It was shown that the calculated probability of failure, that is, that surface temperature exceeds coolant saturation temperature, lies closer to values obtained from the Deterministic Method than to values from the Statistical Method.

The calculated probability value was identified as the probability of success at the instant of reactor startup. It was observed that the probability of success for continued operation might not be the same as the value for startup.

The method gives an improved representation of the probability problem for reactor surface temperatures. However, there is still much to be learned about the various important distribution functions. In the present situation of inadequate knowledge of behavior and distributions of uncertainty factors, all such probability calculations must be regarded as providing only a rough approximation to the true probability of success for a reactor.

INTRODUCTION

The designer of a high power nuclear reactor has to consider not only all the calculated most probable temperatures and heat fluxes, but also the effects of uncertainties in

* University of Toledo, Toledo, Ohio.

all the important variables. That is, the designer must allow for the fact that there are errors in heat transfer correlations, fuel plate loadings, reactor power level measurements, and the like. This results in the reactor being designed to operate with nominal or most probable values considerably less than the safe limits for those values. Of course, this constitutes a penalty, since it results in a reactor which operates at considerably lower power and flux levels than could be allowed if no uncertainties existed.

During the last decade, a number of articles and reports have appeared on the treatment of such uncertainties by statistical methods. Nearly all of them treat the problem of determining the probability that the reactor fuel temperature, either clad surface or fuel centerline, will not exceed a limiting value, such as the saturation temperature of the cooling water. Most of these articles show the application of conventional statistical equations to the calculation of probability of safety for the hot spot in the core. These articles are reviewed in detail by Fenech and Guéron (ref. 1).

Fenech and Guéron developed a Synthesis Method of calculation (ref. 1). It treats the case of a reactor which has tubular cladding with fuel in the form of pellets. This type of fuel allows one to treat each pellet and its associated length of clad as a "characteristic length", where each characteristic length is independent of the others. Then the probability of failure for each pellet in the core is calculated. All the probabilities are multiplied together to get the probability of failure for the whole core, that is,

$$P(\text{failure of core}) = 1 - \prod_{i=1}^n [1 - P(\text{failure of pellet } n)] \quad \text{for } n \text{ pellets} \quad (1)$$

The method is cited as having the following advantages:

- (1) It gives a realistic choice of the smallest region, or spot, that is independent of the other regions in the core.
- (2) It takes into account the whole core rather than just the nominally hottest spot.
- (3) The calculations, though laborious, can be done by hand, and are therefore useful even in the early design stages.
- (4) It does not require assumption of any particular probability distribution for the maximum surface temperature.

The choice of a pellet for a characteristic length may be correct for a tubular fuel form. But this simple a choice is not possible for a plate-type fuel form. Also, the synthesis method fails to recognize that not all uncertainties are purely local in nature. Some quantities, such as reactor power, have error distributions which affect the whole core. Others, such as fuel plate loading, vary over a whole fuel plate. A valid treatment would have to take this into account.

This article will show a method for performing probability calculations for fuel clad surface temperatures of a nuclear reactor. The method will properly treat varia-

bles which vary over the whole core, over one plate, and at one spot. A method of choosing spot size for plate-type elements will be shown, and a more general method discussed. All the methods are chosen for analysis using a small amount of computer time rather than relying on hand calculations.

Fenech and Guéron identified existing methods as the Statistical Method, Deterministic Method, Spot Method, and Synthesis Method. In this report, we preserve this nomenclature and refer to our proposed method as the "Probabilistic Method" for the sake of clarity.

In what follows, the method of combination of probabilities to obtain the probability of success for the whole core is shown. Then the "spot" size derivation is given, followed by its use to obtain probability distributions for the "cell." The uncertainty factors which can exist for an MTR-type reactor are given and an example calculation using the proposed method is shown. This method is compared with the Deterministic Method and with the Statistical Method. Finally, some observations are made about the significance of the values that are obtained by probability calculations of this nature.

SYMBOLS

| | |
|------------|---|
| a | plate thickness |
| C_1, C_2 | constants of integration |
| F_r | radial distribution function |
| h | heat transfer coefficient |
| $I_0(mr)$ | modified Bessel function of first kind, order zero |
| $I_1(mr)$ | modified Bessel function of first kind, order one |
| $K_0(mr)$ | modified Bessel function of second kind, order zero |
| $K_1(mr)$ | modified Bessel function of second kind, order one |
| k | thermal conductivity |
| m | $m^2 \equiv 2h/ka$ |
| $P(A)$ | probability of occurrence of event A |
| $P(A B)$ | probability of A given that B has occurred |
| R | particular value of r |
| RP | reactor power |
| r | radius |

| | |
|--------------------|--|
| T | temperature |
| T_b | coolant bulk temperature |
| ΔT_f | film temperature drop, $T_w - T_b$ |
| $\Delta \bar{T}_f$ | nominal value of ΔT_f |
| $\Delta \hat{T}_f$ | maximum value of ΔT_f |
| ΔT_f^* | arbitrarily chosen member of population of ΔT_f |
| T_{sat} | coolant saturation temperature |
| T_w | surface or "wall" temperature |
| \hat{T}_w | maximum surface temperature |
| X | $\Delta T_f^* / \Delta \bar{T}_f$ |
| α | fraction of heat flux at point due to heat generated within radius R of that point |
| Φ | distribution function |
| $\prod_{i=1}^n$ | accumulated product of n factors |

THE COMBINATION EQUATIONS

To better understand the method to be used here, consider a reactor composed of n cells, each of which operates at the same nominal temperature and each of which has the same distribution of uncertainty in temperature. Each cell is independent of the others. Now, if there is just one factor which varies in a random manner about the mean, and it varies independently in each cell, then

$$P(\text{success for reactor}) = [P(\text{success for cell})]^n \quad (2)$$

by the multiplication rule for probability.¹

However, if this factor is one which varies in the same way over the whole core at once, then the cells are no longer independent, and

$$P(\text{success for reactor}) = P(\text{success for cell}) \quad (3)$$

because all cells will fail at once; in fact, all will run at equal temperature. Evidently

¹"Success" means that the maximum surface temperature is less than the coolant saturation temperature, although this is a conservative limitation for most water-cooled nuclear reactors.

there is a difference in the effect of an uncertainty in a factor which varies over the whole core at once, such as reactor total power, and one which varies locally, such as fuel plate loading.

In the previous example, if both factors are present - that is, uncertainty in both reactor power and plate loadings - then the correct combination of probabilities for l discrete values of the reactor power is

$$P(\text{success for reactor}) = \sum_{i=1}^l \left\{ P(RP_i) \cdot [P(\text{success for cell} | RP_i)]^n \right\} \quad (4)$$

by the decomposition rule for probabilities.

These relations are now extended to the real reactor. Suppose that the probability distribution for the "cell" is known, where the cell is some portion of the length of a fuel plate and each cell is independent of the others. Then

$$P(\text{success for a fuel plate}) = \prod_{i=1}^n P(\text{success for cell } i) \quad (5)$$

There are many fuel plates in a reactor and they have a variety of neutron flux levels and fuel loadings. Thus the probabilities of success of fuel plates differ even at the same reactor power. Then

$$P(\text{success for all fuel plates} | RP) = \prod_{j=1}^m P(\text{success for fuel plate } j | RP) \quad \text{for } m \text{ plates} \quad (6)$$

In some cases this can be considerably simplified. For example, if the vertical power distributions in all the plates are similar in shape, differing only in magnitude, then the radial distribution function, including all the effects that vary between plates, can be described by a probability distribution F_r . Let each plate share the same probability distribution of F_r , so that all plates have equal probability of success. Then

$$P(\text{success of all plates} | RP) = \left[\sum_{j=1}^p P(F_{r_j}) \cdot P(\text{success of one plate} | F_{r_j}) \right]^m$$

for m plates per reactor, p possible values of F_r (7)

Note that this involves no assumptions about the form of the various distributions other than that they are independent over either cell, fuel plate, or core.

SIZE OF A "SPOT"

The purpose of the following derivation is to provide a more logical choice of spot size than used in previous work.

Figure 1 shows a cylindrical section cut through a fuel plate, with the axis of the cylinder normal to the plane of the plate. The plate is considered to be homogenized

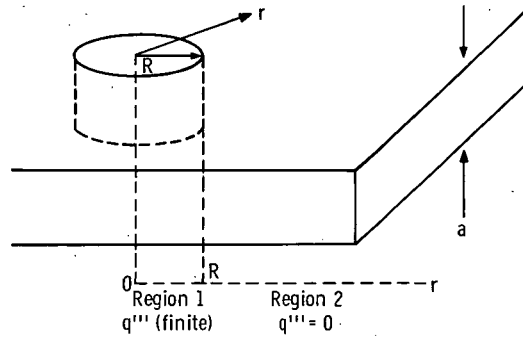


Figure 1. - Fuel plate in vicinity of "spot".

over its thickness. A "spot" is bounded by that radius about a point such that some large fraction (say α) of the heat flux at the point is due to the heat source within the radius. The differential equation describing the temperature distribution radially from the axis of the spot is

$$-k \left(\frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r} \right) + \frac{2h}{a} T = q''' \quad (9)$$

where

$$T \equiv T(\text{plate}) - T(\text{coolant})$$

The general solution is

$$T = C_1 K_0(mr) + C_2 I_0(mr) + \frac{q'''a}{2h} \quad (10)$$

where

$$m^2 \equiv \frac{2h}{ka}$$

Now consider the two-region problem where the heat generation is finite from 0 to R but is zero from R to ∞ . Using the following boundary conditions:

$$\left. \begin{aligned} \frac{\partial T_1}{\partial r} &= 0 \text{ at } r = 0 \\ \frac{\partial T_1}{\partial r} &= \frac{\partial T_2}{\partial r} \\ T_1 &= T_2 \text{ at } r = R \end{aligned} \right\} \quad (11)$$

gives

$$T_1 = C_2 I_0(mr) + \frac{q'''a}{2h} \quad (12)$$

where

$$C_2 = \frac{\frac{-q'''a}{2h} K_1(mR)}{K_0(mR)I_1(mR) + K_1(mR)I_0(mR)}$$

If the spot radius R were extended to infinity, the terms involving the partial differentials in equation (9) would vanish, giving

$$T_1 = \frac{q'''a}{2h} \quad (13)$$

It follows that

$$\alpha = \frac{T_1(0)|_R}{T_1(0)|_\infty} = 1 - \frac{K_1(mR)}{K_0(mR)I_1(mR) + K_1(mR)I_0(mR)} \quad (14)$$

where $T_1(0)|_R$ is the temperature at the axis when the spot radius is R and $T_1(0)|_\infty$ is the temperature at the axis when the spot radius is infinite. For a typical MTR-type fuel element,

$$k \doteq 1.75 \text{ W}/(\text{cm})(\text{K})$$

$$a \doteq 0.15 \text{ cm}$$

$$h \doteq 5.0 \text{ W}/(\text{cm}^2)(\text{K})$$

$$m \doteq 6.0 \text{ cm}^{-1}$$

In figure 2, α is given as a function of R and of mR . For example, α equals 0.9 when the radius is 0.53 centimeter. The diameter of the spot is about 1 centimeter. If a 60-centimeter-long by 6-centimeter-wide plate is divided into 12 lengths, there are about 30 spots per length.

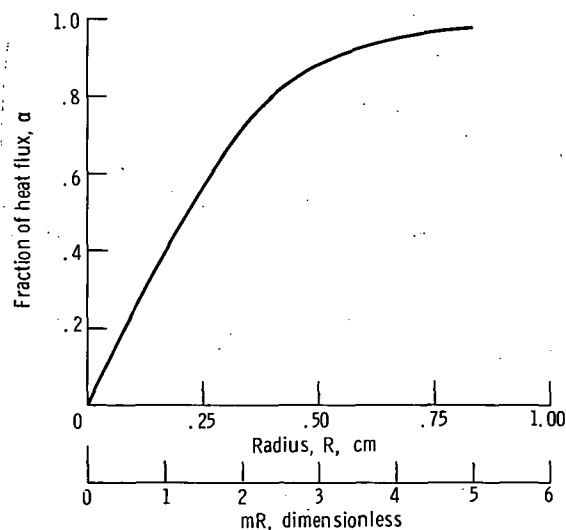


Figure 2. - Fraction of heat flux at a point due to heat generated within radius R of that point as a function of R . (MTR-type fuel elements.)

THE "CELL"

Previously the "cell" was mentioned as being some portion of the length of a plate. In fact, we arbitrarily define it to be some portion of the length of a plate. It extends the width of the plate and is long enough so that there are a conveniently small number of cells per plate. It also is small enough that the nominal heat flux and water temperature are nearly constant in any one cell.² Thus the cell is a convenient concept to use in the calculations, and since it is made up of a known number of "spots" which have known temperature probability distributions, the distribution of maximum temperature of the cell can be calculated.

For example, if there are n spots in the cell, each independent of the others, and each having the same probability distribution of film temperature drop ΔT_f , then

$$P(\Delta T_f \text{ in a cell} < \Delta T_f^*) = \left[P(\Delta T_f \text{ at a spot} < \Delta T_f^*) \right]^n \quad (15)$$

Suppose the distributions for the spots are given as $\Phi(t)$. Then

$$P(\Delta T_f \text{ at a spot} < \Delta T_f^*) = \int_0^x \Phi(t) dt \quad (15)$$

where $x = \Delta T_f^* / \Delta \bar{T}_f$ and Φ is the distribution of $\Delta T_f / \Delta \bar{T}_f$ for the spot. Then

$$P(\Delta T_f \text{ in a cell} < \Delta T_f^*) = \left[\int_0^x \Phi(t) dt \right]^n \quad (17)$$

gives a probability integral table for the cell.

There is still an arbitrariness in the result, arising from the arbitrary choice of α which defines the spot size. Here Monte Carlo methods appear to have a great advantage. That is, construct a two-dimensional finite difference heat transfer model for the cell. Then prepare a number of problems by random sampling from the probability distributions within the cell. In this way, the probability distribution of the maximum film temperature drop factor in the cell can be calculated. Then use the combination equations shown here to build up the probabilities of success for the fuel plates and the reactor core. This work did not include a Monte Carlo calculation, be-

²There may actually be some local flux peaking in the cell. But since the spots are independent, not all spots in the cell have to come from the same physical location. The cell just accounts for a certain number of spots, which are usually close together.

cause the statistical data are not good enough to warrant it; the spot size derivation shown was adequate to do the comparisons and draw the relevant conclusions.

EXAMPLE CALCULATIONS AND COMPARISONS

An example calculation was done using the operating parameters and the uncertainties of a typical light-water test reactor at startup. The operating parameters and the uncertainty factors and their values are listed in table I.

TABLE I. - NOMINAL OPERATING PARAMETERS AND
UNCERTAINTY FACTORS

(a) Nominal operating parameters at startup

| | |
|--|-----|
| Power, MW(th) | 60 |
| Inlet temperature, K | 330 |
| Coolant saturation temperature (at core outlet), K | 440 |

(b) Uncertainty factors

| Region of independence | Factor | Fractional standard deviation percent | Distribution ^a |
|------------------------|---------------------------------------|---------------------------------------|----------------------------|
| Spot | Local fuel density \times thickness | 5 | Normal |
| | Local heat transfer coefficient | 10 | Normal |
| | Flux peak at bottom end of fuel | 4.5 | Normal |
| Plate | Coolant velocity | 6 | Normal |
| | Power distribution | ---- | Rectangular, 1.0 \pm 0.5 |
| | Fuel per plate | 2.5 | Normal |
| | Area per plate | 1.5 | Normal |
| Core | Reactor power | 5 | Normal |

^aThe distributions were all arbitrarily truncated at $\pm 3\sigma$.

The rectangular power distribution is illustrated in figure 3. It was not assumed that the power distribution in the core was uniform; rather, it was assumed that a given plate had equal probability of having any power between 0.5 and 1.5 times the average power. Figure 2 also shows the normal distribution of the product of the coolant velocity, fuel per plate, and area per plate factors. (That distribution was also arbitrarily truncated at $\pm 3\sigma$.) Finally, figure 3 shows the distribution of the product of the factor having normal distribution and the factor having rectangular distribution. The result is the F_r distribution referred to earlier.

Figure 4 shows the probability integral of $\hat{\Delta T}_f / \Delta \bar{T}_f$ for the spot and for the cell when there are 30 spots in the cell. The two curves are related by equation (17). Notice

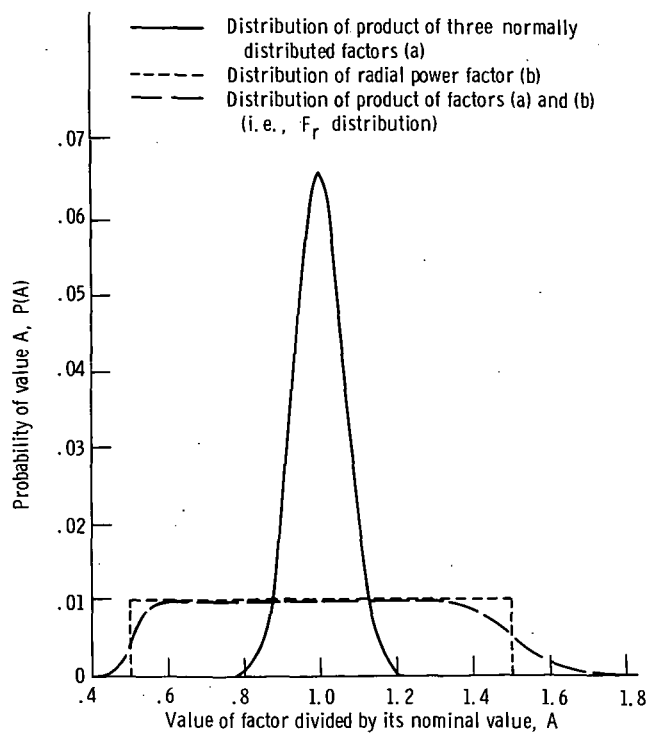


Figure 3. - Probability distributions for factors which vary independently over fuel plates.

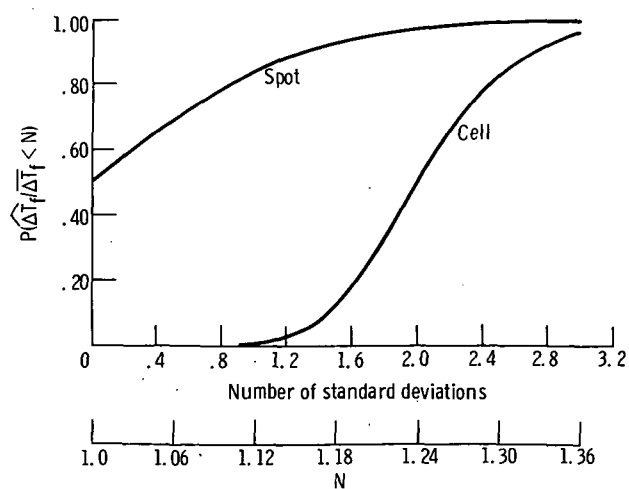


Figure 4. - Probability that $\hat{\Delta T}_f / \Delta T_f$ is less than N in spot and in cell (30 spots per cell).

that there is negligible probability that $\Delta\hat{T}_f/\Delta\bar{T}_f$ will be less than 1.12 for the cell. No value can be greater than 1.36, because the distributions are truncated at $\pm 3\sigma$ (three times the standard deviation).

Figure 5 shows the calculated nominal values of T_{sat} , T_b , and T_w when $F_r = 1.0$ or 1.8 . The numbering of the cells is shown along the abscissa. The value of the ratio $(T_{sat} - T_b)/(T_w - T_b) = 5.75$ at cell 4 when $F_r = 1.0$. The probability that $\Delta\hat{T}_f/\Delta\bar{T}_f$ exceeded 5.75 was zero from figure 4. Thus the probability that $T_w < T_{sat}$ was 1.0 for this cell. The probability that T_w was less than T_{sat} was calculated for

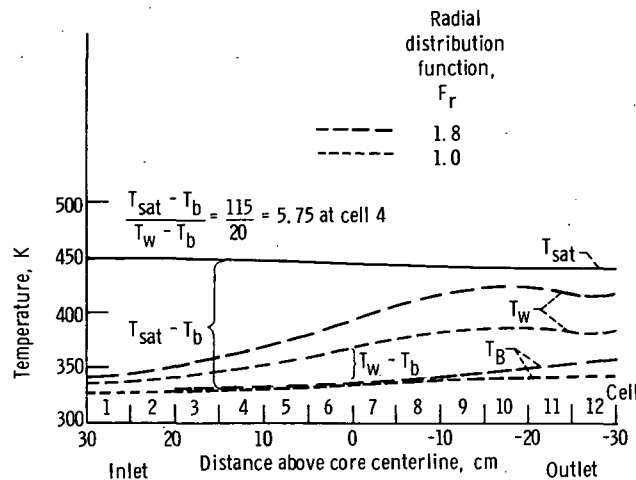


Figure 5. - Temperature as a function of position in channel for two values of F_r . (Reactor at 60 MW.)

each cell and the values combined as in equation (5) to obtain the probability that $T_w < T_{sat}$ for the whole plate. This was done for each value of F_r at each reactor power. Figure 6 shows $P(T_w < T_{sat})$ as a function of F_r at 60 megawatts. This result for each reactor power was used as in equation (7) to obtain figure 7. Figure 7 shows $P(T_w < T_{sat})$ for the whole core as a function of true reactor power. The values given in figure 7 were then used in equation (8) to obtain $P(T_w < T_{sat})$ for the reactor, including the effects of uncertainty in reactor power level.

The uncertainty factors of table I were also used in calculations by the Deterministic Method and by the Statistical Method. (In the Deterministic Method, the maximum possible values are all assumed to occur at once; the $+3\sigma$ values are the maximum possible in this example. In the Statistical Method, the uncertainty in the temperature is calculated only for the nominally hottest spot.)

The results of calculations by the Deterministic Method and by the Statistical Method are compared with the result of the new Probabilistic Method given here in table II. The

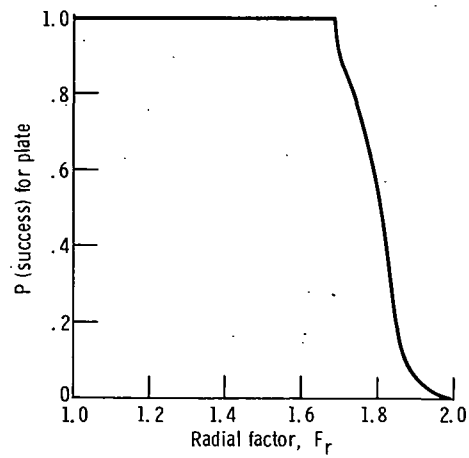


Figure 6. - Probability that $T_w < T_{\text{sat}}$ for one fuel plate as a function of radial factor F_r . (Reactor at 60 MW.)

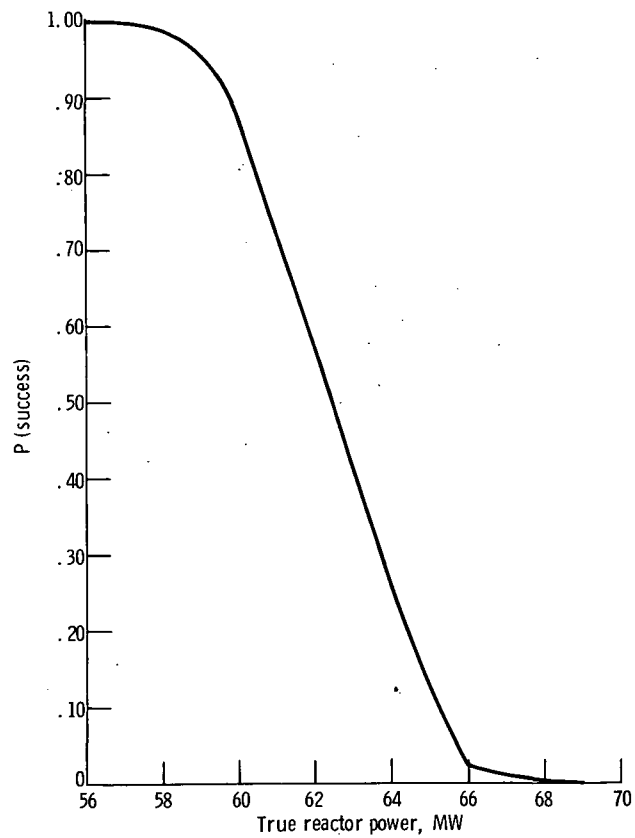


Figure 7. - $P(\text{success})$ for core as a function of true reactor power.

TABLE II. - COMPARISON OF RESULTS USING

DIFFERENT METHODS

| Method | Results | P(success) |
|---|--|------------|
| Deterministic | $\hat{T}_w = 485 \text{ K}$, $T_{\text{sat}} = 440 \text{ K}$ | ----- |
| Statistical | Most probable $\hat{T}_w = 409 \text{ K}$ | 0.9995 |
| Probabilistic | | |
| h varying locally | ----- | .75 |
| h varying over whole core systematically | ----- | .92 |

Probabilistic Method gives a probability of success of 0.75. The Deterministic Method gives a value of 485 K, which is greater than T_{sat} . According to the Deterministic Method, there is zero probability that T_w exceeds 485 K and unknown probabilities for all lesser values.

The Statistical Method gives a most probable \hat{T}_w of 409 K with a standard deviation of 9.7 K. This has a normal distribution such that the probability of success is 0.9995. Thus $P(\text{failure}) = 1 - 0.9995 = 0.0005$ compared to $1 - 0.75 = 0.25$ for the Probabilistic Method. The results differ by a factor of 500. The Deterministic Method in effect gives a $P(\text{failure})$ of 1.0. This result for $P(\text{failure})$ is greater than that of the Probabilistic Method by a factor of 4.0. Thus the Deterministic Method, which is simpler and more conservative than the Statistical Method, also is in better agreement with the Probabilistic Method, at least for this example.

To illustrate the effect of core-wise variation instead of local variation, the Probabilistic Method calculations were redone with one change. The heat transfer coefficient h was assumed to vary systematically over the whole core instead of locally. The result was a $P(\text{success}) = 0.92$ for the core. The probability of failure is thus about three times smaller if h varies core wise rather than locally. This demonstrates the importance of knowledge of the manner of variation of each factor and the importance of proper combination of the different uncertainty factors.

EFFECT OF CONTINUED OPERATION

The probability distribution of maximum surface temperature is a function of time. In an MTR-type reactor, for each core loading, the distributions of fuel and the other variables in the analysis stay about the same as they are at the start of the operating cycle. The neutron flux peaking decreases during the cycle because of control rod withdrawal, so the maximum surface temperature \hat{T}_w tends to decrease with time. This is illustrated in figure 8.

The value 0.75 that we obtained for $P(\text{success})$ of the core is equal to

$$\int_0^{T_{\text{sat}}} f(\hat{T}_w, 0) d\hat{T}_w \quad (18)$$

in figure 8. There is 0.75 probability of starting the cycle in the "safe" zone, say at point P. If conditions stay the same, \hat{T}_w should progress along path A. If some things change in a random manner, \hat{T}_w will follow a "random walk" from left to right in figure 8, with a randomly chosen step toward larger or smaller \hat{T}_w during each step in time. But an accurate construction of this random walk would require knowledge of how rapidly and extensively the variables could change with time, and such knowledge does not now exist.

For the example, it seems reasonable to assume that the calculated value of $P(\text{success})$ equals that for one cycle of operation. But each cycle may be a repeat of the trial because a new core loading is used for every cycle. If so, $P(\text{success})$ for 10 cycles is $(0.75)^{10}$ or about 0.06.

There are two conclusions to draw. First, the value for $P(\text{success})$ calculated by this and most other existing probabilistic or statistical methods is identified as appropriate to the instant of startup. Secondly, the $P(\text{success})$ for any continuing operation may be less than that for the instant of startup. This aspect of the problem must be considered by the designer when using such methods.

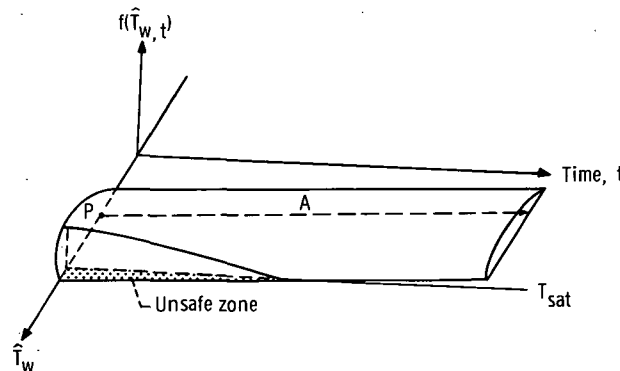


Figure 8. - Probability distribution of maximum surface temperature T_w as a function of time t .

CONCLUSIONS

A method of performing probability calculations for nuclear reactor surface temperatures has been devised. The method gives consideration to the fact that some uncertainties may vary systematically over all the reactor or over some parts of it. The method does not depend on assumption of any particular form of the probability distributions.

The method was used to do an example calculation for an MTR-type test reactor with plate-type fuel elements. It was shown that the calculated probability of failure, that is, that surface temperature exceeds coolant saturation temperature, lies closer to values obtained from the Deterministic Method than to values from the Statistical Method.

The calculated probability value was identified as the probability of success at the instant of reactor startup. It was observed that the probability of success for continued operation might not be the same as the value for startup.

The method gives an improved representation of the probability problem for reactor surface temperatures. However, there is still much to be learned about the various important distribution functions. In the present situation of inadequate knowledge of behavior and distributions of uncertainty factors, all such probability calculations must be regarded as providing only a rough approximation to the true probability of success for a reactor.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, March 14, 1969,
122-29-05-11-22.

REFERENCE

1. Fenech, Henri; and Guéron, Henri, M.: The Synthesis Method of Uncertainty Analysis in Nuclear Reactor Thermal Design. Nucl. Sci. and Eng., vol. 31, 1968, pp. 505-512.

POSTMASTER: If Undeliverable (Section 158
Postal Manual) Do Not Return

"The aeronautical and space activities of the United States shall be conducted so as to contribute . . . to the expansion of human knowledge of phenomena in the atmosphere and space. The Administration shall provide for the widest practicable and appropriate dissemination of information concerning its activities and the results thereof."

— NATIONAL AERONAUTICS AND SPACE ACT OF 1958

NASA SCIENTIFIC AND TECHNICAL PUBLICATIONS

TECHNICAL REPORTS: Scientific and technical information considered important, complete, and a lasting contribution to existing knowledge.

TECHNICAL NOTES: Information less broad in scope but nevertheless of importance as a contribution to existing knowledge.

TECHNICAL MEMORANDUMS: Information receiving limited distribution because of preliminary data, security classification, or other reasons.

CONTRACTOR REPORTS: Scientific and technical information generated under a NASA contract or grant and considered an important contribution to existing knowledge.

TECHNICAL TRANSLATIONS: Information published in a foreign language considered to merit NASA distribution in English.

SPECIAL PUBLICATIONS: Information derived from or of value to NASA activities. Publications include conference proceedings, monographs, data compilations, handbooks, sourcebooks, and special bibliographies.

TECHNOLOGY UTILIZATION PUBLICATIONS: Information on technology used by NASA that may be of particular interest in commercial and other non-aerospace applications. Publications include Tech Briefs, Technology Utilization Reports and Notes, and Technology Surveys.

Details on the availability of these publications may be obtained from:

SCIENTIFIC AND TECHNICAL INFORMATION DIVISION
NATIONAL AERONAUTICS AND SPACE ADMINISTRATION
Washington, D.C. 20546